Benefits and Challenges of the Use of High-Z Plasma Facing Materials in Fusion Devices

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The road to tungsten may be long and tedious…

History of development of filament materials for light bulbs

- metal
- carbon coated carbon
- tungsten

**Driver:** life time  
**Problem:** production

<table>
<thead>
<tr>
<th>Filaments:</th>
<th>1800</th>
<th>1825</th>
<th>1850</th>
<th>1875</th>
<th>1900</th>
</tr>
</thead>
<tbody>
<tr>
<td>platinum</td>
<td></td>
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<tr>
<td>powdered charcoal</td>
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<tr>
<td>carbonised bamboo</td>
<td></td>
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<tr>
<td>carbon fiber</td>
<td></td>
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<tr>
<td>tungsten</td>
<td></td>
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</tbody>
</table>

H.Day | T.Edison | D.Coolidge

SEM 60-watt light bulb filament (75x)
A brief look into history

- **Vacuum compatibility** of PFCs was first priority in early devices → gold plated stainless steel liners in ORMAK

- Low low-Z content → higher edge temperatures & higher performance, better core confinement but higher sputtering source → **impurity accumulation – hollow** $T_e$ in PLT when using W limiters

- Need for low-Z PFM, availability of vacuum grade graphite and benign behaviour under thermal overload → adoption of C PFCs in almost all fusion devices

- Operation with high current high density and/or divertor allows to use refractory (high-Z) metals (low plasma temperatures in contact with PFCs!)
Benefits and Challenges of High-Z PFMs

• Why do we need a substitute for C based materials

• Experiences in present day machines
  – ’High’-Z devices
  – diagnostic for W
  – hydrogen retention
  – W erosion
  – W concentrations and transport
  – behaviour under powerload
  – effect of n-irradiation

• Extrapolation to ITER

• Summary / remaining issues
Why going back to refractory metals?

Motivation to abandon C-based materials in a future reactor
- fuel retention by co-deposition with C
- high erosion of low Z materials
- stability against neutron damage

Challenges for operation of a full high-Z device:
- tolerable impurity level much lower than for low-Z ($c_C \leq 10^{-2}$, $c_W \leq 5 \times 10^{-5}$)
- reliable tokamak operation scenarios
- compatibility of standard & advanced H-mode scenarios with a full high-Z wall
- compatibility of heating methods: ICRF

High-Z devices: TRIAM-1M, FTU, Alcator C-Mod, ASDEX Upgrade
High-Z test PFCs: JET, JT-60U, TEXTOR

Other important constraints
Material properties, change under n-irradiation, diagnostic issues, …
Rationales for plasma facing materials

**Low erosion rates:**
- low power loss by dilution / radiation originating from impurities
- long lifetime of PFCs
- low dust production
- low T co-deposition

![Graph showing sputtering yields for D⁺](image-url)
Rationales for plasma facing materials

**Low erosion rates:**
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- long lifetime of PFCs
- low dust production
- low T co-deposition

**Low atomic number**
- low radiation loss parameter

**Losses through**
- dilution (low-Z): \( n_{DT} = n_e (1 - Z n_Z) \)
- radiation (high-Z): \( \frac{P_{\text{rad}}}{V} = L_z n_z n_e \)
Rationales for plasma facing materials

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**Losses through**

dilution (low-Z) : \( n_{\text{DT}} = n_e (1 - Zn_Z) \)
radiation (high-Z) : \( P_{\text{rad}} / V = L_Z n_Z n_e \)
Boundary Conditions for PFM in a Reactor
Integrated approach necessary

- plasma behaviour: confinement, MHD-stability, operational limits, threshold behaviour
- impurity behaviour: penetration, transport, concentration, radiation, other intrinsic impurities
- erosion, deposition, H-retention, migration
- diagnostic: identification, quantification of influx and density
- technological issues: material properties, machinability, ...
- n-irradiation: embrittlement, swelling, transmutation, activation
Boundary Conditions for PFM in a Reactor
Integrated approach necessary

- Nuclear reactions: $(n,\gamma), (n,2n), (n,p), (n,\alpha)$
  - Produce other elements/isotopes:
    - Change of mechanical properties
    - Radioactive waste
  - Embrittlement by displacements
    - Main concern

- Plasma behaviour:
  - Confinement, MHD-stability
  - Operational limits
  - Threshold behaviour

- Irradiation:
  - Embrittlement
  - Swelling
  - Transmutation
  - Activation

- Impurity behaviour:
  - Penetration, transport
  - Concentration, radiation
  - Other intrinsic impurities

- Erosion, deposition
  - H-retention
  - Migration

- Diagnostic issues:
  - Identification
  - Quantification of influx and density

- Technological issues:
  - Material properties
  - Machinability

- Cracking after thermal shocks
  - Castellation, 'macro-brush' arrangement
Benefits and Challenges of High-Z PFMs

• Why do we need a substitute for C based materials

• Experiences in present day machines
  – ’High‘-Z devices
  – diagnostic for W
  – hydrogen Retention
  – W Erosion
  – W concentrations and transport
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  – effect of n-irradiation

• Extrapolation to ITER

• Summary / Remaining Issues
FTU (ENEA Frascati)

- “all metal” tokamak
  \[ R = 0.93 \text{ m}, \quad a = 0.28 \text{ m}, \quad B_t \leq 8 \text{T}, \quad I_P \leq 1.6 \text{ MA} \]
- first wall
  SS + boronisation
- poloidal limiter
  (until 1994)
  SS, Inconel, TZM, W
- toroidal limiter
  (since 1995)
  TZM (~ 1 m²)
divertor configuration with a complete set of bulk Mo-tiles

one toroidal row of W lamella tiles

B. Lipschultz et al., PSI 2008

- 0.5 mm W (PS) on graphite tiles
- coverage of 90% of the strike zone
- no damage during operation:
  - 800 plasma discharges,
  - heating powers up to 10 MW
  - max. average heat load ≤ 6 MW/m²
Full W ASDEX Upgrade from 2007 on
ASDEX Upgrade (full W since 2007)

W-coating starting with campaign:
- 2003/2004
- 2004/2005
- 2005/2006
- 2007

W coatings on fine grain graphite:
- main chamber, inner divertor:
  - PVD 3-5 µm
- outer divertor
  - VPS 200 µm → PVD 10 µm from 2009 on
JET ITER-like Wall Project (from 2011 on)

Full metal device: inertially cooled
Be main chamber
- bulk limiter and dump plates
- Be PVD coating on Inconel
W divertor /
high power / fluency areas
- tile 5: bulk tungsten
- divertor (except tile 5), main chamber (mainly NBI shinethrough areas):
  10 – 20 µm PVD coating

JET ITER-like Wall Project
Layout of Bulk W Tile (Tile 5)

Full divertor unit
(one row in JET at outer strike point = 48 units
or 96 tiles)

8 stacks of tungsten lamellae
with rear castellation

Inertial cooling +
metals (huge EM forces) +
W is a refractory metal

‘Wedge’ carrier with
deep toroidal
cuts against
eddy current loops

8 feet
rest on the CFC base carrier

Typically 24 bulk-W lamellae
(with sandwiched insulating TZM spacers)

Ph. Mertens (FZ Jülich)
JET ITER-like Wall Project (from 2011 on)

- Beryllium
- Inconel+8μm Be
- Bulk W
- W-coated CFC
- Upper Dump Plate
- Saddle coil protection
- Mushrooms
- Poloidal Limiters
- Inner Wall Cladding
- Restraint Ring Protections
- Normal NBI Inner Wall GL’s
- Normal NBI IW Cladding
- Saddle Coil Protections
- LH + ICRH Protection
- Divertor
- B&C tiles
- Re-ionisation Protections
- Magnetic covers
- Inner Wall Guard Limiters
- Div. tile 5
# Devices for plasma irradiation: linear plasma simulators

### PISCES-B Devices

<table>
<thead>
<tr>
<th>Parameter</th>
<th>PISCES-B</th>
<th>Confinement Devices</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ion flux (m(^{-2}) s(^{-1}))</td>
<td>(10^{23})</td>
<td>(10^{23} - 10^{24})</td>
</tr>
<tr>
<td>Ion energy (eV)</td>
<td>20-300 (bias)</td>
<td>10-300 (thermal)</td>
</tr>
<tr>
<td>Heat flux (MW/m(^2))</td>
<td>1-10</td>
<td>1-10</td>
</tr>
<tr>
<td>(T_e) (eV)</td>
<td>2-40 (thermal)</td>
<td>1-100 (thermal)</td>
</tr>
<tr>
<td>(n_e) (m(^{-3}))</td>
<td>(10^{17}-10^{19})</td>
<td>(10^{18}-10^{20})</td>
</tr>
<tr>
<td>Impurity fraction (%)</td>
<td>0.03-10</td>
<td>1-10</td>
</tr>
<tr>
<td>B (Gauss)</td>
<td>200-1000</td>
<td>10,000</td>
</tr>
<tr>
<td>Pulse length</td>
<td>continuous</td>
<td>10-30 sec</td>
</tr>
<tr>
<td>Fluence/disch.(m(^{-2}))</td>
<td>up to (10^{27})</td>
<td>(10^{24} - 10^{25})</td>
</tr>
<tr>
<td>Target materials and coatings</td>
<td>C,W,Be,Li, C,W,Be,etc.</td>
<td>(any unirradiated)</td>
</tr>
<tr>
<td>Surface Temp(°C)</td>
<td>RT-1100</td>
<td>RT-500</td>
</tr>
<tr>
<td>Plasma species</td>
<td>H,D,He</td>
<td>H,D,T,He</td>
</tr>
</tbody>
</table>

Example: Pisces-B

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R. Doerner et al., UCSD
Devices for power loading of (W) PFCs

QSPA plasma parameters (ELMs):
- Heat load: $0.5 - 2 \text{ MJ/m}^2$
- Pulse duration: $0.1 - 0.6 \text{ ms}$
- Plasma diameter: $5 \text{ cm}$
- Magnetic field: $0 \text{ T}$
- Ion impact energy: $\leq 0.1 \text{ keV}$
- Electron temperature: $< 10 \text{ eV}$
- Plasma density: $\leq 10^{22} \text{ m}^{-3}$

steady state:
plasma generators,
ion beams, e beams

transients:
e beams, plasma guns, quasi stationary plasma accelerators

A. Zhitlukhin, 17th PSI, 2006
Benefits and Challenges of High-Z PFMs

• Why do we need a substitute for C based materials
• Experiences in present day machines
  – ',High'-Z devices
  – diagnostic for W
  – hydrogen retention
  – W erosion
  – W concentrations and transport
  – behaviour under powerload
  – effect of n-irradiation
• Extrapolation to ITER
• Summary / remaining issues
Spectroscopic diagnostic of fusion plasmas
S/XB method for influx measurements

Schematic view of processes involved in W-flux measurements

**Increase of** $T_e, n_e$

**Prompt**

**Re-deposition**

**Emission** of a photon

**W** plasma facing component

**W** plasma

**W**$^{2+}$, **W**$^{1+}$, **W**

**XB**

**X**

**S**$^{`}$, **S**

**Sputtering**

**Line of sight**

**Prerequisite:**

recombination negligible

- Influx

$$\Gamma_{z+1} = \int_{\text{los}} n_e n_z S \, dx$$

- Photon flux

$$\Gamma_\gamma = \int_{\text{los}} n_e n_z XB \, dx$$

$$\Rightarrow \Gamma_{z+1} / \Gamma_\gamma \approx S / (XB) (x_0)$$

S: ionisation, X: excitation

B: branching ratio
S/XB for W I (400.9 nm)

Calculations of S/XB for W I (400.9 nm)
I Beigman et al. PPCF 49 (2007) 1833

- ionisation rate
  ATOM code calculations (lowest configurations)

- excitation rate:
  - semi-empirical v. Regemorter formula (complicated coupling scheme + configuration mixing)
  - corona approximation: only excitation from ‘ground’ state
Spectroscopic diagnostic of fusion plasmas
Ionisation shells in the central plasma

ionisation equilibrium governed by
Coronal approximation

\[ \frac{\partial}{\partial t} n_Z + \nabla \cdot \dot{\Gamma}_Z = n_e (n_{Z-1} S_{Z-1} + n_{Z+1} \alpha_{Z+1} - n_Z S_Z - n_Z \alpha_Z) \]

weak influence of plasma transport on shell structure

\[ \dot{\Gamma}_Z = D_Z \nabla n_Z + v_Z n_Z \]
Spectroscopic diagnostic of fusion plasmas
Impurity concentrations from LOS measurements

Comparison of measured $I_M$ and calculated $I_C$ intensities

$$I_C = \frac{1}{4\pi} \int_\ell h\nu n_x n_e \langle \sigma v_e \rangle dl$$

- $n_x$ density of impurity in ionisation state $x$
- $n_e$ electron density
- $\langle \sigma v_e \rangle$ excitation rate coefficient

$$n_x = C_{imp} \cdot f_x \cdot n_e$$

- $f_x$ fractional abundance of the impurity ionisation state $x$
- $C_{imp}$ impurity concentration

$C_{imp}$ only valid within the emission shell!
W Spectroscopy in the VUV and SXR
Revision of ionization equilibrium

Deduced fractional abundance versus temperature different discharges: symbols different spectral lines: colours

Use of CADW ionisation rates (S.D. Loch, PRA 2005) and adjustment of recombination rates allows good description of emissions of W^{24+} - W^{48+}

Th. Pütterich (PPCF 50 2008 085016)
W Spectroscopy in the VUV and SXR
Revision of ionization equilibrium

standard 'ADPAK'
ionisation/recombination rates

CADW
ionisation rates
ADPAK recombination rates
(adjusted to experiment)
Modelled W emission (ADAS) @ different temperatures

Th. Pütterich, PPCF 50 (2008) 085016

Int. ITER Sum. School, Aix en Provence, June 22-26, 2009
R.Neu
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Strongly reduced D retention after transformation to W PFCs in ASDEX Upgrade

deposition areas: strong reduction of D co-deposition with C
erosion areas: slight increase due to diffusion in W

consistent with laboratory results and particle balance measurements

K. Sugiyama et al., subm. to NF
(Surprisingly) large D retention in Mo observed in C-Mod

retention ~ independent of density when normalized to divertor ion fluence

- implantation of ions drives retention
- reaches 1-2%, >> that predicted for Mo
- stronger fluence dependence than from laboratory data

reason(s) for discrepancy not yet clear:
- trap creation by impinging ions
- reduction of recombination by impurities

B. Lipschultz et al., PSI 2008 and NF49 (2009) 045009
Co-deposition ratio from laboratory and linear devices

Dependent on temperature, energy, flux ratio:

\[
\begin{align*}
(D+T)/C &= (2.0 \cdot 10^{-2}) E^{-0.43} \left( \Gamma_{(D+T)}/\Gamma_C \right)^0 \exp^{(2268/T)} \\
(D+T)/\text{Be} &= (5.82 \cdot 10^{-5}) E^{1.17} \left( \Gamma_{(D+T)}/\Gamma_{\text{Be}} \right)^{-0.21} \exp^{(2273/T)} \\
(D+T)/\text{W} &= (5.13 \cdot 10^{-8}) E^{1.85} \left( \Gamma_{(D+T)}/\Gamma_{\text{W}} \right)^{0.4} \exp^{(736/T)}
\end{align*}
\]

Example:

\[T = 500\text{K}, \quad \Gamma_{(D+T)}/\Gamma_{\text{imp}} = 100\]

\[(D+T)/C = 0.6\]

\[(D+T)/\text{Be} = 0.04\]

R. Doerner, UCSD
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W erosion yield

• W sputtering yields 100-1000 times smaller than for C, but strongly dominated by low-Z intrinsic impurities
• hints for prompt redeposition of W
• very small migration into main chamber
• even larger yields (> $10^{-2}$) in TEXTOR

K. Krieger JNM 266-269 (1999) 207
Increase of limiter W sources and W concentration with ICRH in AUG

ICRF heating causes strong increase of limiter erosion and W concentration

- W sputtering induced by accelerated particles in rectified sheath
- limiter source much more 'efficient' than divertor source (> 5 times, depending on discharge parameters)

similar results for Mo sputtering at C-Mod

correlation also found for increased limiter source in radial scans

R. Dux, JNM 390-391 (2009) 858
ELM Cycle at Low Divertor Density

- Divertor temperature between ELMs: \( \approx 20\text{eV} \)

- Considerable influx also inbetween ELMs

- ELMs contribute \( \leq 50\% \) to the W-influx

R. Dux, JNM 390-391 (2009) 858
ELM Cycle at Higher Divertor Density

divertor temperature is low after the ELM
⇒ erosion yield is lower inbetween ELMs
⇒ ELMs contribute > 50% to W-influx

R. Dux, JNM 390-391 (2009) 858
ELM resolved W erosion in the outer divertor of AUG

**ELM: edge localized modes**

→ periodic release of energy and particles at the edge

‘hot’ divertor: $T_e \sim 20$ eV  
- similar erosion profiles during ELMs and between ELMs  
- ELM contribution $\sim 50$

‘cold’ divertor: $T_e \sim 6$ eV  
- between ELMs erosion much smaller, highest erosion far in the scrape of layer → ‘semi’-detached  
- ELM contribution $> 80$

erosion mainly by intrinsic low-Z impurities

R. Dux, JNM 390-391 (2009) 858
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Control of high-Z transport

Prevent too high W-concentration ($C_W < 5 \cdot 10^{-5}$) and W-accumulation

Very centre of plasma
Suppression of neo-classical accumulation

Confinement region
- Turbulent transport
- Weak impurity gradients

H-mode
Edge Transport Barrier
Control W-influx

Control of ELM frequency
(reduce inward transport of W)
- Gas puffing
- Input power
- other methods

Central power
NBI, ECRH, ICRH
Impurity behaviour
Alcator C-Mod / FTU

General behaviour ($c_{\text{Mo}}$)

• (very) high during low density limiter phase
• strongly decreasing with density
• at high density 2-3 lower in divertor phase
• (also from comparison with FTU)
• 2-5 times higher in H-Mode compared to L-Mode
• reduction by 2-10 through boronisation

$c_{\text{Mo}}$ in C-Mod
usually **no high-Z accumulation**, but only small fraction (< 10%) of PWI (particle/ power flux) on high-Z components

**W accumulation for**
- ohmic discharges above critical plasma density
- high level of (edge) radiation

central high-Z contamination depends strongly on transport (RF heating beneficial)

V. Philipps et al., EPS 1995, p.321
W behaviour in AUG
Reduction of W content by increasing ELM frequency

- ELM frequency depends strongly on gas puffing, distance to H/L threshold and machine conditions
- Larger heating power or ELM pacemaking keep impurity content low

Frequent ELMs necessary to clamp impurity content
W behaviour in AUG
Suppression of central impurity peaking

Central wave heating strongly suppresses impurity peaking

- ECRH more efficient than ICRH (e-heating vs. powerflux)
- Reduction of peaking by increased anomalous transport
- Moderate reduction of total confinement

![Graphs showing suppression of impurity peaking](image-url)
Suppression of W accumulation in AUG

central W accumulation connected to electron density peaking
(can be controlled by central heating and/or gas puff)

neoclassical transport decreases with Z small increase of anomalous transport sufficient

A. Kallenbach et al., NF 49 (2009) 045007
Suppression of W accumulation in JT-60U

- W accumulation provoked by counter NBI
- Strong suppression of W accumulation by central heating
  (increase of turbulent transport / destabilization of sawteeth)
- AUG results confirmed

T. Nakano et al., 22nd IAEA FEC 2008, EX/P4-25
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Tungsten melt layer behaviour

- Liquid W jet moves up with velocity of about 1.5 m/s
- Material loss: 2.85 g of W removed from the erosion channel in 1s
- Motion of molten W and outward propagation of the jet are due to the thermo-emission current
- Variation of the depth of the erosion channel in pol. direction probably due to additional heat transfer by liquid metal flow
W behaviour under high heat loads

- recrystallisation starts above ~1200°C (lower fracture toughness)
- no enhanced erosion found close to melting
- re-solidified surfaces are prone to increased power loads

Under transient heat loads
- development of cracks (fatigue, below melt-temperature)
- melt layer movement and losses due to jxB force, plasma pressure, ...
  (e-beam, plasma gun, QSPA experiments: very difficult to adjust to ITER parameters)

TEXTOR test limiter experiments with W-macrobrush structures

G. Sergienko et al.,
JNM 390-391 (2009) 858
Damage thresholds for CFC and W under ELM-loads

CFC
- negligible erosion

<table>
<thead>
<tr>
<th>Energy density* E / MJm(^{-2})</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
</tr>
<tr>
<td>0.5</td>
</tr>
<tr>
<td>1.0</td>
</tr>
<tr>
<td>1.5</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Heat flux factor P \cdot \sqrt{\Delta t} / MWm(^{-2}) s(^{1/2})</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
</tr>
<tr>
<td>20</td>
</tr>
<tr>
<td>40</td>
</tr>
<tr>
<td>60</td>
</tr>
</tbody>
</table>

W
- negligible damage

 associations

- melting of tile edges
- melting of tile surface
- droplets bridging of tiles
- cracking of pitch fibres
- crack formation

- mitigated ELMs in ITER

* \(\Delta t = 500\) \(\mu s\)

J. Linke

Int. ITER Sum. School, Aix en Provence, June 22-26, 2009

R. Neu
During the first shot droplets ejected mainly from the edges of the tiles. As a result of edge smoothing and bridging of gaps the droplet ejection was reduced and mass losses were decreased.

A. Zhitlukhin et al., SRC RF TRINITI, Troitsk
Bridge formation @ \( E \geq 1 \text{ MJ/m}^2 \)

\begin{align*}
\text{W3,R3, 20 exposures} & & \text{W3,R3, 50 exposures} & & \text{W3,R3, 100 exposures} \\
\text{W4,L3, 10 exposures} & & \text{W4,L3, 20 exposures} & & \text{W4,L3, 50 exposures} \\
\end{align*}

\[ w = 1.0 \text{ MJ/m}^2 \]
\[ w = 1.6 \text{ MJ/m}^2 \]
JUDITH electron beam experiment ($\Delta t = 5$ ms); major cracks, microcracks and surface modification.

- $\Delta T = 606^\circ C$
- $\Delta T = 1131^\circ C$
- $\Delta T = 1697^\circ C$

- Bulk temperature (oC)
- Absorbed power density (GW.m$^{-2}$)

- Brittle

- Recrystallization

- J. Linke, FZ Jülich
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Neutron irradiation effect on thermal conductivity

J. Linke,
Thermal fatigue testing of a W macrobrush module irradiated in the HFR-Petten

irradiation condition: 200°C – 0.1 dpa (in W)

loading condition: 1000 cycles at 10 MW/m²

CuCrZr
Cu
WLα₂O₃

Cu

CTE = 5·10⁻⁶ K⁻¹

J. Linke,
Thermal fatigue testing of W monoblock mock-ups

<table>
<thead>
<tr>
<th></th>
<th>W-monoblock</th>
<th>W-monoblock</th>
<th>W-lamellae design</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>ENEA</td>
<td>CEA</td>
<td>Plansee AG</td>
</tr>
<tr>
<td><strong>unirradiated</strong></td>
<td></td>
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<tr>
<td>1000 x 14.5 MW/m²</td>
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<td></td>
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</tr>
<tr>
<td>1000 x 9.6 MW/m²</td>
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<td></td>
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</tr>
<tr>
<td>1000 x 18.0 MW/m²</td>
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</tr>
<tr>
<td><strong>0.1 dpa</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$T_{irr} = 200°C$</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1000 x 10.0 MW/m²</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>100 x 13.7 MW/m²</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1000 x 17.9 MW/m²</td>
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</tr>
<tr>
<td><strong>0.6 dpa</strong></td>
<td></td>
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<tr>
<td>$T_{irr} = 200°C$</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>1000 x 10.0 MW/m²</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1000 x 13.7 MW/m²</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1000 x 18.0 MW/m²</td>
<td></td>
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</tr>
</tbody>
</table>


no failure observed!
T retention in (neutron) induced traps

- traps produced during ITER lifetime neutron fluence: \( \sim 0.005/W @ 200^\circ C, \) 
  \( \sim 0.0001/W @ 500^\circ C. \)

- 200\(^\circ\)C: trapped D is limited by slow kinetics, i.e. permeation. 
  (uptake rate and D concentration in solution, is three orders of magnitude smaller than predicted by model based on diffusion and surface recombination!)

- 40\(^\circ\)C: smaller trapping due to slower kinetics

- 500\(^\circ\)C: smaller trapping due to annealing of damage.

\[ \Rightarrow \text{low T inventory in W from trapping due to neutron damage in ITER.} \]
Benefits and Challenges of High-Z PFMs

• Why do we need a substitute for C based materials
• Experiences in present day machines
  – 'High'-Z devices
  – diagnostic for W
  – hydrogen retention
  – W erosion
  – W concentrations and transport
  – behaviour under powerload
  – effect of n-irradiation
• Extrapolation to ITER
• Summary / remaining issues
Wall loads in future confinement experiments

<table>
<thead>
<tr>
<th></th>
<th>W7-X</th>
<th>ITER</th>
<th>reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>heat flux FW / MWm$^{-2}$</td>
<td>$&lt;0.2$</td>
<td>$1$</td>
<td>$&lt;1$</td>
</tr>
<tr>
<td>heat flux divertor / MWm$^{-2}$</td>
<td>$&lt;1$</td>
<td>$20$</td>
<td>$\approx 5-20$</td>
</tr>
<tr>
<td>VDEs / MJm$^{-2}$</td>
<td>?</td>
<td>$60$</td>
<td>-</td>
</tr>
<tr>
<td>disruptions / MJm$^{-2}$</td>
<td>$\approx 10$</td>
<td>$\approx 10$</td>
<td>-</td>
</tr>
<tr>
<td>ELMs / MJm$^{-2}$</td>
<td>?</td>
<td>$&lt;1$</td>
<td>?</td>
</tr>
<tr>
<td>neutron fluence / dpa</td>
<td>?</td>
<td></td>
<td>?</td>
</tr>
</tbody>
</table>

- thermal fatigue
- thermal shock
- degradation, embrittlement
Extrapolation to ITER: Safety Limits

Number of discharges to reach ITER safety limits:

- Hot dust limit: 6kg
- T in Be limit: 100000 kg
- Hot dust limit: 230 kg

The all metal / W solution would be best in respect of safety limits.

J. Roth et al., JNM 390-391 (2009) 1
Extrapolation to ITER: Edge W concentrations

W erosion and edge plasma contamination in ITER from **DIVIMP calculations for several B2-E backgrounds** (edge transport not fully understood!)

W conc. remain under $2 \cdot 10^{-5}$ **for any coverage level by W PFCs in ITER** and high density operation (weakly influenced by seeding, $D_{an}$ & parallel flows)

K. Schmidt, JNM 363-365 (2007) 674
Extrapolation to ITER: Central impurity transport

No W accumulation expected in ITER if 
\[ D_{an} \geq D_{neo}, \quad (v/D)_{an} \text{ not increasing with } Z \]
(as predicted, see C. Angioni et al., PPCF 49 (2007) 2027)

calculations using

\[ Q = 10, \quad P_{NBI} = 40 \text{ MW}, \quad U_{\text{loop}} = 75 \text{ mV} \]

\[ D_{\text{neo}}, \quad v_{\text{neo}} \text{ from NEOART} \]

\[ (v/D)_{an} \text{ from fit to GLF23} \]

\[ D_{an} \text{ varied} \]

R. Dux et al., 20th IAEA FEC 2004, EX/P6-14
Summary

• D-retention in metals low (lab. exp., AUG), but high retention in C-Mod not yet resolved

• ‘destructive’ transients (large ELMs, disruptions) not accessible in present day machines (except large ELMs in JET, disruptions in C-Mod)

• erosion of high-Z materials mainly by low-Z impurities – transients and accelerated particles (ICRF) play significant role

• main chamber sources dominate plasma impurity density although much lower than divertor source

• AUG achieves similar performance as in boronized C device, using
  – sufficiently high particle transport in the plasma centre by central heating
  – flushing of pedestal by sufficiently high ELM frequency

• safety limits (T retention / Dust / Erosion) best for full metal / full W ITER

• extrapolation of edge/central transport seems favourable for ITER
Remaining Issues and Extrapolation to ITER and DEMO

- mixed materials effect (He, low-Z) on surface morphology / D retention
- effect of divertor damage / behaviour of melt layers under tokamak conditions
- optimization of plasma edge / antenna design (reduction of parasitic electrical fields) for reduction of W source during ICRF
- effect of pellet ELM pacemaking and RMP ELM suppression:
  - evolution of edge plasma parameters / W source
  - penetration into confined plasma / flushing
⇒ JET ILW and lab experiments combined with modelling may close some of the gaps to ITER in the near future

**DEMO: step in plasma physics much smaller compared to step in PWI!**
- PFC: full W (or and W and steel)
- high PFC temperature necessary: good for annealing of defects and T-retention but low margin for transients, large T diffusion
- high n-fluence: dpa ~100 times larger as in ITER
## Properties of candidates as PFM

<table>
<thead>
<tr>
<th></th>
<th>Be</th>
<th>CFC</th>
<th>W</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>atomic number Z</strong></td>
<td>4</td>
<td>6</td>
<td>74</td>
</tr>
<tr>
<td><strong>max. allowable concentration in the plasma</strong></td>
<td>~3 %</td>
<td>~2 %</td>
<td>~20 ppm</td>
</tr>
<tr>
<td><strong>thermal conductivity ( \lambda ) [W/mK]</strong></td>
<td>190</td>
<td>200 ... 500</td>
<td>140</td>
</tr>
<tr>
<td><strong>melting point [°C]</strong></td>
<td>1285</td>
<td>&gt;2200 (subl.thr.)</td>
<td>3410</td>
</tr>
<tr>
<td><strong>coefficient of thermal expansion [10(^{-6}) K(^{-1})]</strong></td>
<td>11.5</td>
<td>~0 **</td>
<td>4.5</td>
</tr>
<tr>
<td><strong>n-irradiation behaviour</strong></td>
<td>swelling</td>
<td>decrease in ( \lambda )</td>
<td>activation</td>
</tr>
</tbody>
</table>

\* CTE copper = 16 \( \cdot 10^{-6} \) K\(^{-1} \)

\* NB31 in pitch fiber direction
Thermal conductivity of different plasma facing materials

![Graph showing thermal conductivity vs temperature for different materials](image-url)
Optimization of ICRH in AUG

ITER: substantial amount if ICRH, high power densities at antenna:
⇒ erosion of Be first wall / limiters and W baffles must be kept low

- further investigations on acceleration mechanism (near field/far field)
- optimization of operational conditions (density, phase, …)
- reduction of box currents / electrical fields by improved antenna design